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OUTLINES OF THE FRENCH R&D PROGRAMME FOR THE DEVELOPMENT OF HIGH AND VERY HIGH TEMPERATURE REACTORS

Philippe Billot – CEA **Dominique Hittner – AREVA NP**

Philippe Vasseur – EdF

ABSTRACT

A R&D programme has been launched in partnership between CEA, AREVA NP and EDF, addressing the needs of the ANTARES programme of AREVA meant at developing an indirect cycle flexible modular HTR operating at 850°C for electricity generation and/or heat production for industrial processes and exploring the potential of this type of reactor for higher performances in terms of fuel burn-up and temperature (VHTR objective), in particular for application to hydrogen production. Demonstrating the viability and the performance of the HTR/VHTR requires meeting a number of significant technical challenges for each following project:

Design and safety analysis for the HTR/VHTR needs the development of computational tools. This requires extending the existing codes and developing coupled calculations in reactor physics, thermo-fluid dynamics, mechanics as well as the validation of the codes based on experimental databases.

The HTR/VHTR fuel identified as TRISO-coated particle must exhibit excellent performance in operating and accidental conditions. The main R&D goals are: the development of manufacturing process, the fuel qualification based on irradiation and heating safety tests and the development of fuel behaviour models including performance data, failure particle prediction and long-term integrity of the coating.

The material and component technologies that have to be investigated in normal and accident conditions for ANTARES and VHTR objectives are those of respectively, medium temperature (~ 450-650°C) for vessel structures and cold internals, high temperature (~ 650-950°C) for primary circuit, intermediate heat exchanger and very high temperature (~ 1000-1650°C) for reactor core structural elements. In order to demonstrate that materials have adequate performance over long service life under impure helium environment and constraints, the research programme focuses on micro-structural and mechanical property data, long-term irradiation behaviour, corrosion, modelling and codification of design rules as well as qualification of components in representative helium test loop.

The key issues of the R&D programme on hydrogen production technologies include: developing and optimizing the thermo-chemical water splitting processes of the “sulphur family”, evaluating alternative thermo-chemical hydrogen generation processes, advancing the high temperature electrolysis process, defining and validating technologies for coupling the reactor and the process plant at different scale.

This paper addresses the R&D work that has been launched since 2001. It is envisioned that the main results to establish the viability issues are expected by 2010 and will benefit the cooperation within the Generation IV International Forum and in Europe with the Euratom 6th and 7th Framework Programmes.

INTRODUCTION: THE ANTARES PROGRAMME

The R&D programme on HTR technology presented in this paper, jointly implemented by the CEA, AREVA NP and EdF, is meant at supporting the development of an industrial multipurpose HTR prototype through the ANTARES programme of AREVA NP. CEA, AREVA NP and EdF are also strongly involved in the project RAPHAEL (coordinated by AREVA NP) of the Euratom 6th Framework Programme, presented in another paper [1], which brings important complementary R&D contributions to ANTARES. The French partners are also involved in other international cooperation actions, in particular in the discussions of the Generation IV International Forum, which hopefully will lead to a fruitful international R&D programme.

The ANTARES programme and its objectives are also presented in this conference [2], but its main characteristics are briefly reminded here.

The flexibility requirement is the main driver for selecting ANTARES design options. The objective is to be able to adapt the reactor design to different types of uses, from electricity production to industrial heat applications at different levels of temperature with possible combination of different uses (e.g. cogeneration), with limited changes. This requirement of ***flexibility*** leads to choose an indirect cycle, which allows decoupling of the nuclear reactor from different possible applications. The key specific component for the indirect cycle is the intermediate heat exchanger (IHX), which is rather challenging: it has to transfer from the primary to the secondary circuit a power of several hundred MW with high efficiency, to operate at very high temperature (core outlet temperature) and to be compact enough to be enclosed into a single vessel (with multi-loop tubular IHX design as a back-up). The potential of different plate type designs for compactness is presently examined. In order to maximize the power, an annular core with hexagonal block type fuel assemblies is selected, with the fuel composed of standard UO_2 TRISO coated particles. A maximum use is made of the inherent safety feature of the HTR and its fuel (highly negative temperature coefficient, very large thermal inertia, leak tightness of the TRISO particles to fission gas up to very high temperatures…) in order to simplify the safety design.

The main reference design parameters are shown in Table 1. Variations are still possible around these values.

Table 1: ANTARES reference design parameters

Thermal power (MW)	600
Core outlet temperature (°C)	850
Core inlet temperature (°C)	400
Primary fluid	He
Pressure (bar)	70
Fuel enrichment	~ 15%
Fuel discharge burn-up (GWd/tHM)	150
Secondary side IHX outlet temperature (°C)	800
Secondary fluid (for electricity generation)	80% N_2, 20% O_2

MAIN R&D NEEDS FOR ANTARES

As mentioned above, the IHX is a challenging component, in particular for plate designs, as there is no industrial experience with the size and operating conditions aimed at. The temperature (850°C) and environmental conditions (inescapable presence of impurities in the helium atmosphere that may induce drastic lifetime reduction of the thin IHX plates through carburisation, decarburisation or oxidation reactions) require considering other types of materials (nickel base alloys) than industrially used for plate IHX. The existing databases on mechanical properties, ageing and corrosion of these materials have to be widely complemented in order to be able to select the most appropriate material in 2007 and then to qualify the selected material. The fabricability of the IHX (forming plates, assembling their bundle and welding it with headers) with such materials must be demonstrated and finally the performances and the lifetime of the different possible designs will have to be assessed in order to select the most appropriate design by the end of 2008 and to be qualified afterwards.

For testing the IHX and other components (valves, hot gas duct, circulator…), test loops must be developed. In the beginning simple ones are sufficient for selecting the best design options on the basis of "separate effect" tests; later bigger and more complex facilities, allowing tests at representative scale and operating conditions, will be necessary for qualification.

In order to operate the reactor with a higher core inlet temperature than for PWRs (for improving efficiency and lowering the peak fuel temperature) and improving margins for accident conditions, another material than PWR vessel steel has been selected, the modified 9Cr1Mo steel. This material must be qualified for ANTARES operating conditions and the feasibility of vessel fabrication assessed (procurement of large ingots, forging, thick welding).

The graphite grades used in past HTR projects are no more available. A selection is to be made among present industrial grades proposed by manufacturers, for which there is no experience of operation in HTR conditions and the selected grade must be qualified.

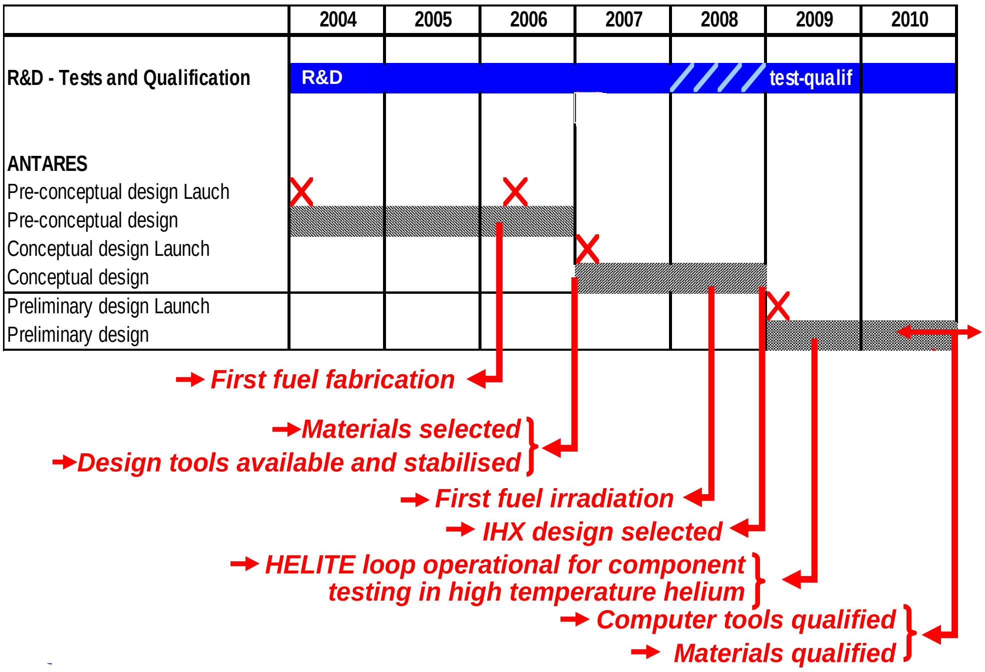
For control rod cladding, which will operate in high neutron flux and high temperature conditions, a solution with C-C composites is looked for as metallic materials appear not to be able to satisfy the requirements. The most appropriate composite in terms of cost, mechanical properties, irradiation and corrosion behaviour is to be selected and qualified.

Europe has acquired in the past a large experience of manufacturing and operating in HTRs high quality TRISO fuel. The high performance of such fuel in normal and accident conditions has been fully demonstrated (qualification up to ~ 80 GWd/tHM and to a fuel temperature ~ 1100°C + evidence of significant margins for higher performance). Such high quality fuel manufacturing techniques must be recovered and even improved in order to reach the higher performance target of ANTARES. Such higher performance must be demonstrated.

The existing design computer tools are not satisfactory for designing a modern HTR. Design tools inherited from former HTR developments are obsolete: computer codes progressed in the last 20 years, both in terms of physical modelling (even if the understanding of HTR physics acquired from past experience remains useful) and in terms of architecture. On the other hand, modern tools used for designing other types of reactors are not adapted to specific features of HTR physics. Anyway tools have to be qualified for application to ANTARES configuration and operating conditions. Nevertheless the situation is specific to each area:

- Reactor physics – the IAEA benchmarks have shown [3] that the physical modelling of specific features of HTR reactor physics have to be improved (neutron streaming, very heterogeneous fuel structure, neutron spectrum…) and qualified,
- Thermo-fluid dynamics – the state-of-the-art codes are able to model the complex flow situations in HTR, but uncertainties in this modelling have to be better known and reduced for specific critical situations (mixing in the lower reactor plenum, core bypass, flow distribution in the IHX headers…) through tests of representative mock-ups,

Figure 1: main ANTARES R&D need milestones



- System transient analysis – existing codes used for PWR transient analysis (like MANTA in AREVA NP and CATHARE in CEA) must be adapted by introducing models of specific ANTARES components (core, IHX) and gas properties and then qualified,
- Fuel performance – taking into account the specificity of the HTR fuel, new modelling is to be developed and qualified,
- Mechanical design – the capability of mechanical design tools for modelling specific HTR materials (e.g. graphite) and configurations (e.g. the stack of hexagonal blocks of the core) must be assessed and specific models developed and qualified (e.g. core seismic analysis).

Moreover, it should be emphasized that R&D results have a key role in the selection of design options, in design optimisation and prototype licensing. Therefore it is not only important to obtain the requested R&D results, but also to obtain them in due time. From that standpoint, the tight schedule for satisfying R&D needs, as illustrated in figure 1, is to be pointed out.

The different parts of the R&D programme meant at answering ANTARES needs are presented in the next parts, as well as results already obtained.

MATERIALS R&D

The objectives of the material R&D are:

- To select the most appropriate material for each component within a list of candidates which are selected among the materials that, on the one hand, have the potential to satisfy the project design and safety requirements and on the other hand, can be qualified and be available on an industrial basis within the time frame of the project, which means that advanced materials with no industrial experience will be considered only for longer term applications,
- To acquire the comprehensive set of data on the selected materials required for design and licensing and to qualify them for their application in the project.

This R&D is closely connected with the effort of codification of materials for HTR/VHTR applications, in which AREVA NP is actively involved (in particular in the frame of the ASME working groups) and with the definition of design rules.

Vessel material

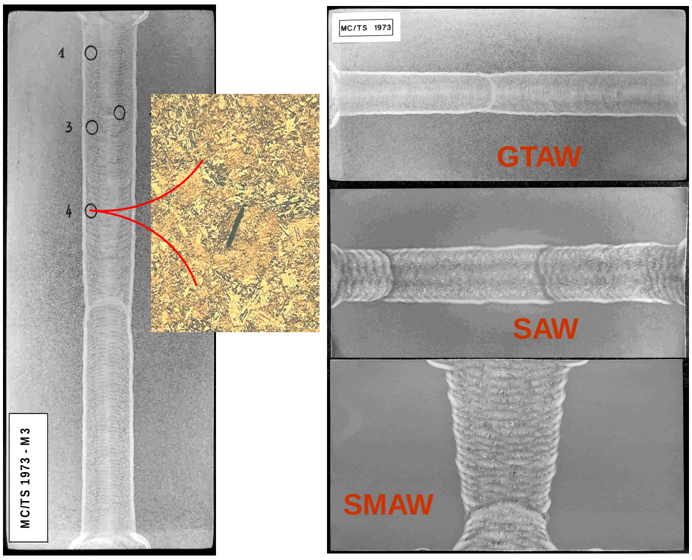
Two options are considered [4]:

- The "hot" vessel option, in which the vessel is kept in normal operation at a temperature in the range 400-450°C, for which the pre-selected candidate is the modified 9Cr1Mo steel,
- The "cold" vessel option, in which the vessel is kept at a temperature close to 370°C, allowing the use of SA 508 (PWR steel), with possible an extension of the ASME code case covering the accident conditions (about 540°C for less than 1000 hours).

Mod. 9Cr1Mo can also be considered for internals, but, as other ferritic steels, modified 9Cr1Mo cannot withstand temperatures exceeding 800°C without a risk of austenisation leading to a drastic loss of mechanical properties, which, taking into account temperatures reached in accident conditions, restricts the possibility of using mod. 9Cr1Mo to the coldest internal structures.

For SA 508, AREVA NP relies on its very large database, and, even if some additional data might perhaps be required on the behaviour of this steel in accident conditions (to be confirmed), the present R&D programme is focused on mod. 9Cr1Mo (except tests performed with both materials for verifying the absence of environmental effects in impure helium atmosphere and for emissivity measurement).

Knowing that the behaviour (and in particular the creep properties) of mod. 9Cr1Mo and of thick welded joints of this



(a) (b)

Figure 2: Mod. 9Cr1Mo thick welding: (a) initial GTAW test with hot cracking (<0.3 mm); (b) optimised welds

material under irradiation is being addressed in the European programme with already very positive results [4], the R&D performed in France is focused on

- Weldability,
- Determination of the negligible creep limit, creep behaviour in the range (400-500°C) and creep fatigue interaction.

After first attempts of GTAW welding in which hot cracking appeared (figure 2 (a)) and could not be simply eliminated, a programme for adapting welding processes to mod. 9Cr1Mo was launched in AREVA NP technical centre. A simple and fast test procedure was developed by CEA for comparing the tendency of different filler materials to hot cracking (Varestraint test, see figure 3). Finally nearly satisfactory welding (no hot cracking and good mechanical performance) was obtained for 3 processes (GTAW, SAW and SMAW), required for different operations in the workshop and on the site (figure 2 (b)). Progress is still necessary for a 4th process, GMAW.

Figure 3: Varestraint test

Due to discrepancies between the creep behaviour of mod. 9Cr1Mo for moderate temperatures (425-550°C) in RCC-MR and ASME, databases had to be revisited and complemented through additional tests (figure 4) [5]. Moreover in order to avoid having to monitor creep damages trough the whole lifetime of the reactor, the vessel should operate in the negligible creep domain and therefore the negligible creep temperature limit of this material, which is known to be in the range 400-450°C, should be determined more accurately. With the help of the newly acquired creep strain data and available database, it is expected that this limit should be about 425°C for the full reactor lifetime (420 000 hrs). However extended evidence is necessary for validating this preliminary result.

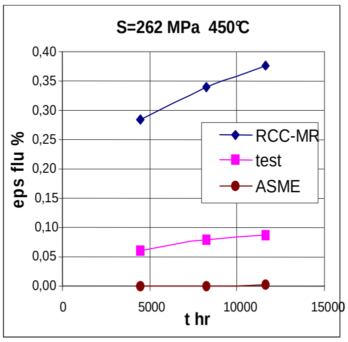


Figure 4: Creep properties of mod. 9Cr1Mo at moderate temperature

For design in accident conditions, it is considered that the existing file with data up to 788°C is sufficient, at least for first design phases, as far as base material is concerned.

Another point to improve in the material data file of mod. 9Cr1Mo is the creep-fatigue interaction diagram. The diagram presently given by the subsection NH of ASME III is very severe and leads to a drastic reduction of the admissible number of cycles as soon as some creep damage is present even at low level. A reassessment of the fatigue relaxation data at 525 °C and 550 °C is available from the EFR programme with less penalizing behaviour (figure 5). Complementary creep-fatigue and fatigue-relaxation tests are necessary for confirming this trend.

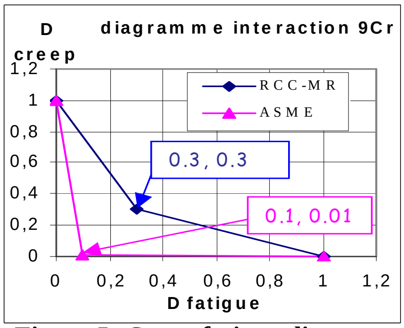


Figure 5: Creep fatigue diagram, comparison ASME/RCC-MR

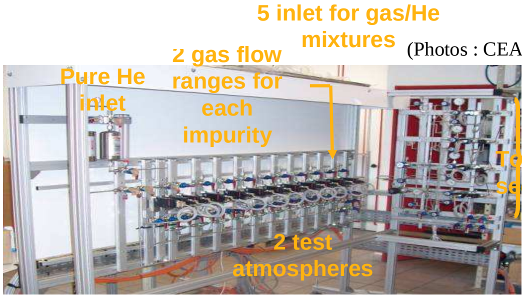
In a second phase, if the decision is made to use mod. 9Cr1Mo for ANTARES, a comprehensive data file will be elaborated on this material and the final qualification will be made on a representative thick forged ring.

High temperature metallic materials

A metallic material that can operate at outlet core temperature (850°C for ANTARES, with exploratory studies for higher temperature) is necessary for the intermediate heat exchanger (IHX). Metallic materials that can operate at intermediate temperature levels between the inlet and outlet core temperatures must also be considered for internal structures. A pre-selection of 2 nickel base alloys, alloy 617 and alloy 230, has been made for operation at the highest temperature level on the basis of existing data and a final selection of materials will be made on the basis of an R&D programme [6] addressing the following aspects:

- Mechanical properties,
- Ageing,
- Environmental effects (corrosion in impure helium atmosphere, nidriding in secondary atmosphere – mixture of nitrogen and helium),
- Fabricability (forming and welding).

Already a significant amount of data has been obtained on mechanical properties, ageing, environmental effects, but it is not possible to decide between the two materials yet, the result of their comparison depending on the particular property under consideration. The final selection, expected



(Photos : CEA Saclay)

Figure 6: Two of the CEA corrosion loops dedicated to ANTARES programme

early next year will follow from a compromise between performances of the material with regard to different properties, the relative importance of which for ANTARES is presently being assessed.

Alloy 617 and alloy 230, will not be discriminated by tensile and creep properties, which are very close above 800°C, though alloy 230 has better tensile properties at lower temperatures. As for Charpy impact values, they are much lower for alloy 230 than for alloy 617.

Nickel base alloys are particularly sensitive at high temperature to oxidation, carburisation and decarburisation caused by inherent impurities (O_2, H_2, H_2O, CO, CO_2, CH_4 are the main ones) of the helium atmosphere in a primary circuit with large quantities of graphite. For addressing these uniform corrosion, creep/corrosion, fatigue / corrosion and creep-fatigue / corrosion issues, a large range of test parameters have to be screened and therefore many test facilities are necessary: 7 complementary helium loops with controlled impurities are already operated or are in development in AREVA, CEA and EdF (see for instance figure 6). It has already been shown that alloy 230 is less sensitive to corrosion than alloy 617 in a representative atmosphere of normal operation conditions (figure 7) and that, at very high temperature (above ~ 965°C), the protective oxide layer formed at lower temperature volatilises, which fixes an upper bound to the operating temperature of these materials [7].

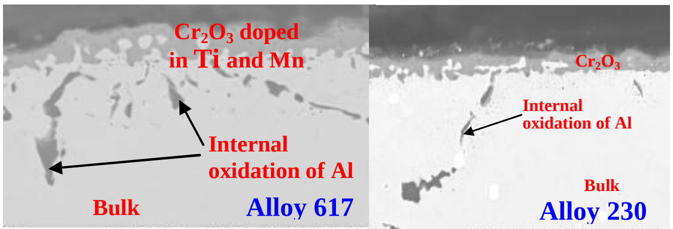


Figure 7: comparative corrosion behaviour of alloy 617 & alloy 230 in representative He atmosphere (950°C, 800 hr)

Regarding fabricability issues of plates made of the pre-selected materials, the main results obtained until now concern:

- The laboratory scale feasibility of plate-fin assembly through brazing,
- The feasibility of engraving straight or slightly wavy channels through fast machining,
- The feasibility of diffusion bonding, even if the process still requires optimisation.

For longer term possible applications with higher performances (in particular higher temperature), exploratory investigations on advanced materials (oxide dispersed steels) are also performed.

Graphite

On the basis of a screening of grades proposed by graphite vendors for HTR application performed in particular in the frame of the European HTR material Programme, a pre-selection of two grades, PCEA from GRAFTECH and NBG-17 from SGL has been made for ANTARES.

A programme has been launched with CEA on these two grades, including comprehensive characterisation, behaviour in operating conditions (irradiation, oxidation in normal and accident conditions and emissivity measurement). Irradiation will start in the CEA OSIRIS reactor in 2008. An irradiation test with in situ creep measurement is considered in a later phase.

Composites

Composites are considered for some high temperature applications under irradiation, most particularly for control rod cladding. Presently an exploratory phase is underway in order to identify the most favourable structures in terms of mechanical properties, irradiation (first fibers and then the composites themselves) and corrosion behaviour. It will be followed by a programme for qualifying the selected composites.

THE DEVELOPMENT OF THE INTERMEDIATE HEAT EXCHANGER (IHX) AND OTHER COMPONENTS

In a first step, during the conceptual design phase, different plate IHX concepts [2] have to be discriminated for their performances in terms of heat transfer capability (the objective being to transfer the maximum power with the maximum efficiency within the minimum volume) and lifetime in HTR operating conditions and the different factors which can affect the performances and the lifetime of the IHX have to be understood in order to be able to optimise the design. This can be done through “separate effect” tests in different tests loop which will provide separately the conditions that can affect the IHX behaviour:

- The heat transfer capability of the current heat exchange part will be measured in the helium loop HEFUS 3 of ENEA that provides representative thermo-fluid dynamics conditions (full pressure, helium flow rate 0.35 kg/s, but temperature only ~ 500°C) in an assembly of a limited number of plates. These tests will take place in 2007, in the European project RAPHAEL [1]. Tests in a helium loop operated by OKBM (Russia) are also under consideration with a funding from the European Commission in the frame of ISTC.

- The homogeneity of flow distribution from the header to all the plates of an IHX module, which is a key factor of the IHX efficiency, will be assessed in cold conditions in air in the large flow rate loop PAT (flow rate 1.3 kg/s, maximum pressure 1.5 MPa, laboratory temperature) of EdF (figure 8). The tests would take place in 2007 and 2008.
- The thermo-mechanical performance of different IHX concepts will be assessed through tests of small IHX mock-ups manufactured with the same materials and assembling methods than the actual IHX, submitted to representative temperature gradients and transients. For that purpose, the CLAIRE loop of CEA (figure 9), which has already been used for testing direct cycle recuperator mock-ups at about 500°C, is presently upgraded to be capable of providing an air flow (maximum 0.2 kg/s, 0.6 MPa) at 950°C and representative thermal transients (cool-down ~ 300°C in 5s, heat-up ~ 300°C in 120s). The tests will be performed in 2008.

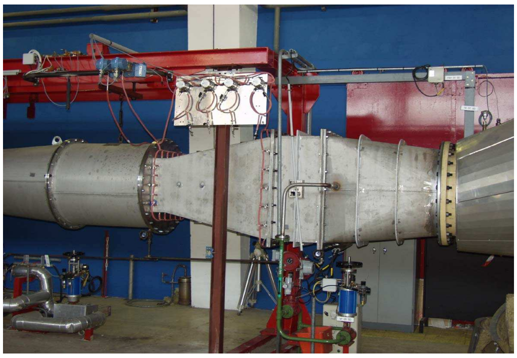


Figure 8: the PAT loop

- During the first phase, possible degradation of the IHX thermo-mechanical performances, caused by chemical interactions of materials with impurities of the primary helium environment and with nitrogen in the secondary atmosphere, is assessed only through elementary tests, in which only simple specimens of IHX materials and of their joints – and not representative mock-ups – are exposed to high temperature (but low pressure) and representative atmospheres (see materials R&D).

The second step, during the preliminary design phase will be the validation and optimisation of the selected IHX design through more representative tests on the HELITE loop, which will provide a first level of integration:

- Full temperature and pressure and representative temperature and pressure transients,
- Representative helium atmosphere (with controlled impurities),
- But limited flow rate.

The HELITE loop (1MW, 950°C, 8 MPa, He in primary circuit and N_2 + He in secondary circuit, flow rate 0.4 kg/s), the detailed design of which is completed (figure 10), will start operation in 2009 in CEA Cadarache.

Due to its limited flow rate, only small mock-ups can be tested in this loop. The final qualification of the IHX will require, on top of the conditions provided by HELITE a much larger flow rate (~ 5kg/s), necessary for tests representative of the behaviour of an IHX module. The loop that will provide these conditions will have a power in the range 10 to 20 MW, but is not precisely defined yet.

The planned large helium loops will be suited for the qualification of other key components (hot gas duct, circulator, isolation valves…), but in the present phase, as they are not available yet, dedicated test benches are necessary for initial design validations. A mock-up of the circulator will be tested in air, most likely in a test facility of the manufacturer. For study of friction and wear issues related to component operation in the helium atmosphere, two helium tribometers have been built, one in CEA Cadarache (figure 11) and the other one in the Technical Centre of AREVA NP. The insulation performance of different types of hot gas duct thermal barriers and their behaviour in depressurisation transients are assessed in the test bench HETIMO (figure 12) in CEA Cadarache. The test bench HETIQ (figure 13) is used for testing the performance of helium leak tight sliding seals.

FUEL DEVELOPMENT

Fabrication

A first step of development of the fabrication process has been completed mainly in the European HTR fuel programme of the Euratom 5th Framework Programme [8]:

- Very small scale UO_2 kernel fabrication by GSP process,
- Coating on dummy kernels.

Kernels and coated particles with specifications similar to those of the past German TRISO particles (sphericity, layer thickness…) could be obtained.

Based on this experience, a joint CEA-AREVA NP laboratory scale facility (CAPRI facility) for UO_2 TRISO particle and compact fabrication has been built and has started operation in 2005 (figure 14) [9]. The first UO_2 TRISO particles with depleted U and the first compacts with ZrO_2 dummy particles have been obtained. Compacting tests with depleted U coated particles will be performed before the end of 2006. In 2007, the fuel (both coated particles and compacts) with enriched U will start to be manufactured for the first experimental irradiation.

In a later phase, an industrial pilot line with the key components (sol-gel reactor and coater) identical to the components of the industrial production plant will be developed and the industrial fabrication process will be qualified.

Exploratory studies are performed on advanced fuel fabrication (UCO kernel, ZrC coating).

Characterisation

Characterization methods are developed [10] [11] with two objectives:

- Allowing a better knowledge of the micro-structural characteristics of the HTR fuel and of the correlation of these characteristics with the fabrication parameters on the one hand and with the performance of the fuel in operating conditions on the other hand, in order to

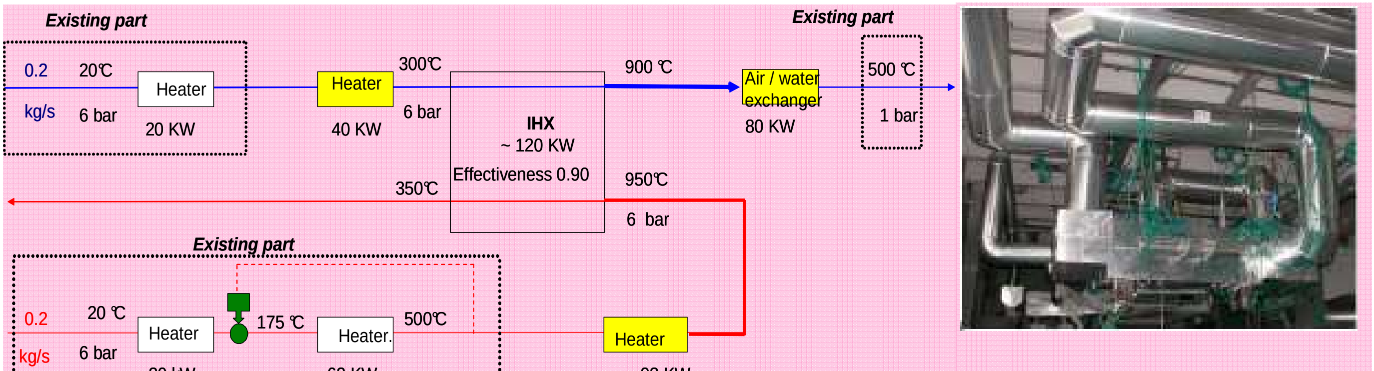
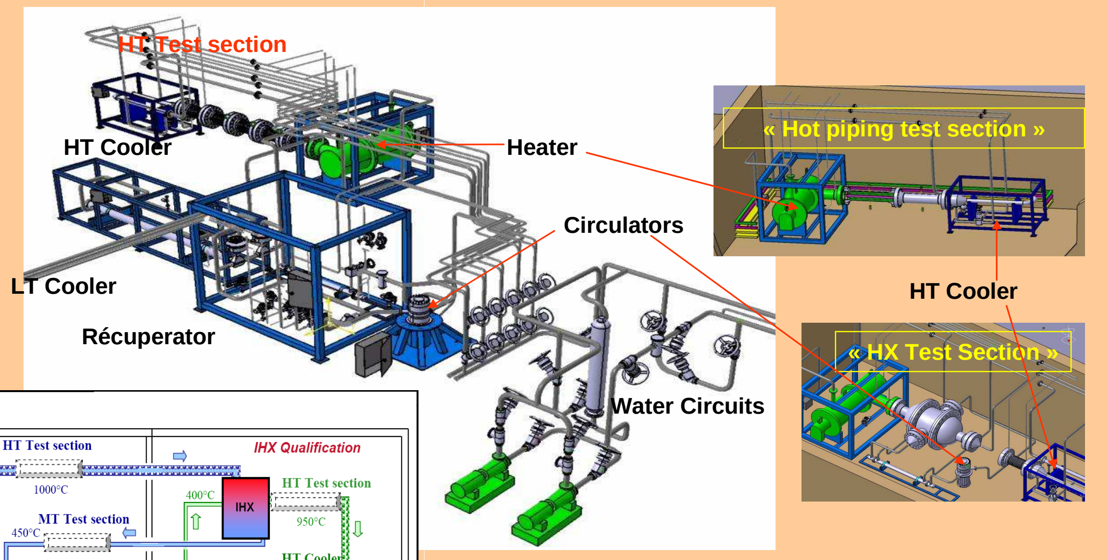


Figure 9: the CLAIRE loop



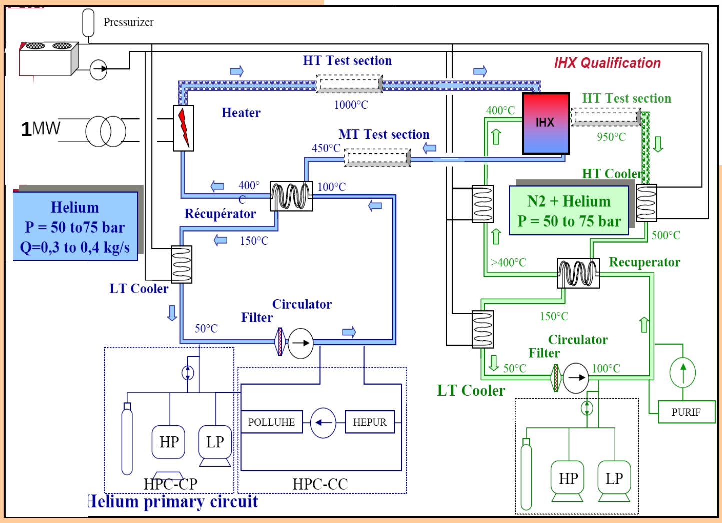


Figure 10: The HELITE loop

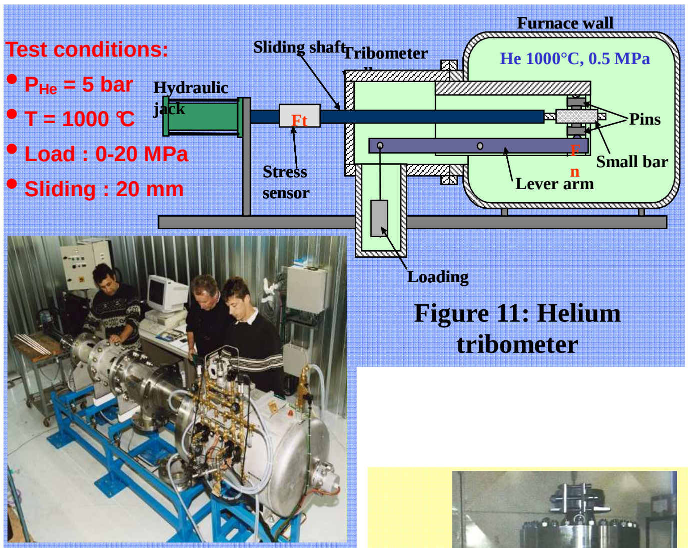
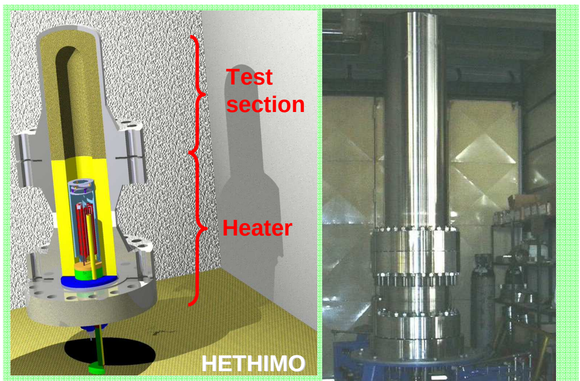
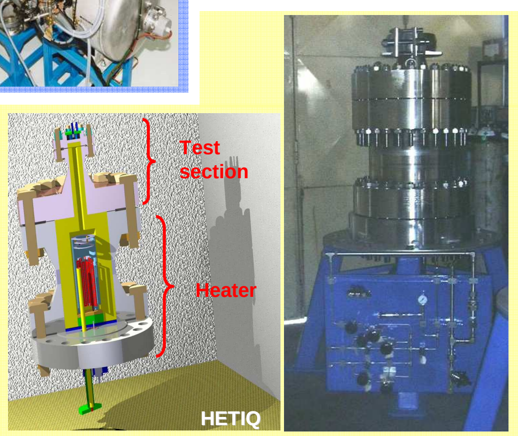


Figure 11: Helium tribometer



Tests conditions:
• P = 100 bar, T = 1000℃
• ΔP/ Δt = 20 bar/s

Figure 12: HETHIMO (HE THermal Insulation MOck up)



Tests conditions:
• P = 100 bar, T = 500℃
• DN100

Figure 13: HETIQ (HE TIghtness Qualification)

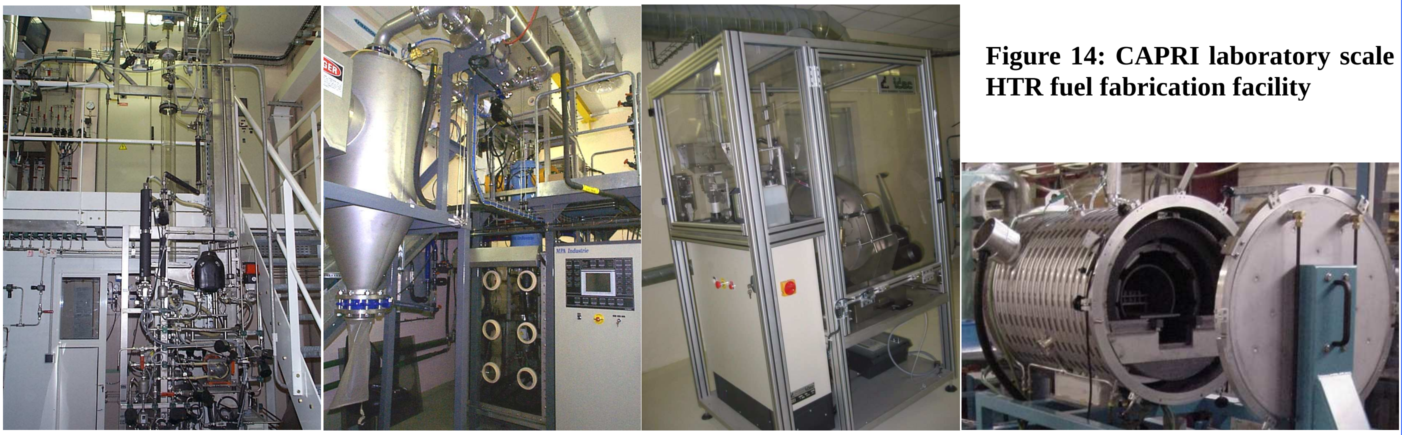


Figure 14: CAPRI laboratory scale HTR fuel fabrication facility

Kernel fabrication *Coater* *Tumbling drum* *Fuel compact furnace*

Coated particle fabrication: GAIA facility (CEA Cadarache) *Compact fabrication (CERCA Romans)*

improve the mastery of the fabrication process and the quality of the product,

- Defining efficient industrial quality control methods that can provide in a cost effective way a clear demonstration of the very high reliability of the first barrier for the fuel from the industrial manufacturing plant.

Qualification

A facility for irradiation of the fuel from the CAPRI facility with on-line fission gas monitoring is being developed (figure 15) and will be set up in the OSIRIS reactor (CEA, Saclay). A programme of 2 irradiation campaigns is planned for the qualification of the fuel from the CAPRI facility [12]:

- SIROCCO1, at low burn-up, for assessing through short irradiation in 2008, the quality of CAPRI fabrication and comparing it to the quality of German fabrication, German reference particles being introduced in some of the compacts irradiated in the test,
- SIROCCO2, at high burn-up, for assessing the performance of the fuel from CAPRI in ANTARES operating conditions (15% FIMA, fuel surface temperature 1200°C and 1000°C).

An extensive programme of pre-irradiation characterisations, post-irradiation examinations and safety test is planned together with the irradiation campaigns.

The qualification of the fuel manufactured in the industrial pilot line, which can be considered as representative of the industrial fuel, will follow.

WASTE MANAGEMENT

The fuel cycle initially considered is once through with direct disposal of irradiated fuel. Nevertheless the important volume of irradiated graphite to be disposed off and the need to fulfil Generation IV goals, in particular sustainable development, lead to study also the possibility of a more or less complete separation of carbonaceous materials from fuel (compacts from blocks, coated particles from compact matrix, kernels from coating layers) and then to consider separate managements for the separated waste flows, with different options of disposal or recycling of the different types of wastes [13].

The legacy from past programmes and the first results obtained by CEA and in the frame of the European programmes are not yet sufficient for demonstrating the industrial feasibility of the different options, but allow anticipating promising prospects.

The study of the direct disposal option was started at the European level during the 5th Framework Programme [14] and the study is presently continued in the frame of the RAPHAEL project [1] as it involves very long term leaching experiments.

The key issue for separation of carbonaceous wastes from the fuel, which needs substantial R&D effort, is at the level of the compact and of the coated particle. Investigations are made on 2 techniques for de-structuring of compacts: acoustic waves and pulsed currents. First encouraging results have been obtained by CEA with this last technique (figure 15).

Principle scheme Before treatment After treatment

Figure 15: pulsed current de-structuring

As for irradiated graphite, its activity results from activation of initial impurities and adsorption of fission products.

Activation calculation and feedback from the French graphite moderated reactors show that the main source of activation is ^{14}C generated from nitrogen adsorbed in graphite. Instead of separating it after irradiation, avoiding nitrogen adsorption in graphite pores would be the most effective solution. This is not so much an issue to be addressed through R&D but a challenge for fabrication and handling.

Decontamination processes are being studied in particular within the European HTR programme. They would

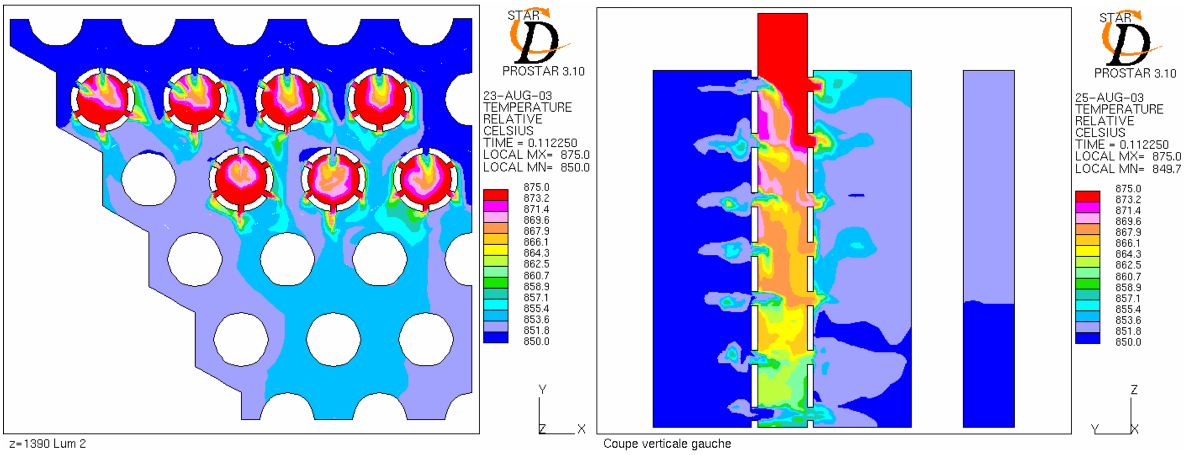


Figure 16: Modelling of the helium flow in the lower plenum with STAR-CD

allow cleaning graphite not only from adsorbed fission products, but also from activated impurities.

COMPUTER CODE DEVELOPMENT AND QUALIFICATION

From the late 1990's, international benchmarks and analyses performed in AREVA NP and CEA allowed identifying the main HTR specific modelling development and qualification needs. In order to consolidate and make the identification of needs more systematic, PIRT type analyses are performed. On the basis of this approach, a programme of code development and qualification programme has been defined. The main features of this programme are described hereafter.

Reactor physics

A specific HTR calculation scheme, NEPHTIS, adapted to HTR prismatic block type core calculation, has been developed based on the spectral code APOLLO 2 [15] and the core calculation tool CRONOS 2 [16] of CEA. Some specific features of the HTR cores in general and the block type cores in particular, such as the double heterogeneity of the HTR fuel (particles and compacts), the neutron spectrum discretisation and the neutron streaming through empty channels of the core blocks, have been the object of dedicated modelling. NEPHTIS is presently operational and used for design, even if some developments are still ongoing.

Due to the scarcity of relevant experimental data, the qualification of NEPHTIS is obtained not only through comparison with such data, but also through code to code comparison (both with Monte-Carlo calculations performed with MCNP and the CEA TRIPOLI code and with reference APOLLO 2 full core deterministic transport calculations.

Now, due to the fact that in NEPHTIS only prismatic block type core calculation capability has been developed, the qualification of NEPHTIS against experiments is based on a two-step process in order to benefit not only from block type core experimental data, but also the pebble bed ones.

In the first step, Monte Carlo codes are qualified against all available experiments performed on both prismatic and pebble-bed HTR configurations at criticality (zero power): HTTR, HTR-10, Fort Saint Vrain (FSV), PROTEUS, ASTRA, HFR, etc. For experiments involving spent fuel, in particular the isotopic analysis of a pebble irradiated up to very high burn-up in HFR, which is planned in RAPHAEL project [1], MONTEBURNS (coupling of MCNP and ORIGEN for evolution calculation) will be used. The purpose of this step is to qualify the so-called "qualifier" reference code systems. As they do not make any modelling approximation, the Monte Carlo code results depend mainly on the accuracy of the master nuclear data libraries in the physical range studied. The main purpose of this first approach is therefore to validate them.

In the second step, qualification of the deterministic industrial calculation scheme NEPHTIS can be established. When available, NEPHTIS is directly compared to experimental data available on specific configurations of prismatic reactors (HTTR, FSV). For all other configurations, qualification of the deterministic industrial scheme is completed by code-to-code comparisons with the reference code systems (MCNP for calculations based on fresh fuel and MONTEBURNS for those involving spent fuel) qualified in the first step [17].

Based on this approach, a reasonable level of validation of the NEPHTIS has been obtained for the needs of pre-conceptual and conceptual design phases.

On top of the qualification with available data an analysis of the need and feasibility of a dedicated ANTARES critical test has been launched.

Thermo-fluid dynamics

State-of-the-art CFD codes (like the commercial code STAR-CD used in AREVA NP and the CEA TRIO code) can be easily used for HTR needs with only minor developments (e.g. the addition of an equivalent porous medium model or of a core physics model in STAR-CD). The main issue is the qualification of the modelling for specific ANTARES applications in order to assess the uncertainties attached to the calculation results. The checking of the qualification of basic physical models and elementary cases is ongoing. In order to qualify the code for calculation of complex cases, critical issues (like the core conduction cool down in a loss of flow accident, the mixing of cold bypass streaks with hot helium leaving the core in the lower reactor vessel plenum (see a calculation performed with STAR-CD in figure 16), the core bypass calculation, the distribution of flow from the collectors to the different plates of the IHX, the reactor vessel pit cooling through the Reactor Cavity Cooling System, etc.) have been identified: specific mock-ups will be tested when the design will be sufficiently defined.

System analysis

Both the AREVA NP PWR transient analysis code MANTA, which will be used for engineering studies, and the

CEA CATHARE code, which will be used for reference calculations, have been adapted to HTR applications, by introducing laws for gas and mixture of gas circulation (physical properties, friction coefficients, heat transfer correlations) and models for HTR specific components (circulator, IHX, core, etc.).

Qualification with code to code benchmarks (MANTA with CATHARE from CEA, LEDA from EdF, RELAP 5, REALY 2 from General Atomics, ASURA from Mitsubishi [18]) and comparison with available experimental data (HTR-10 transients, EVO loop, HEFUS 3 loop (ENEA), South African micro-model…) is partly completed and provides a sufficient confidence level for design studies.

Fuel performance

A code dedicated to HTR fuel performance, ATLAS, is being developed by CEA [19]. It allows coupling the modelling of the thermo-mechanical behaviour (see for instance a calculation of the amoeba effect in figure 17) in of the coated particle and the calculation of fission product releases to be coupled within a single code. A statistical approach allows modelling the behaviour of thousands of particles in a fuel element and then thousands of fuel elements in the whole core [20]. The fuel materials properties used in the code are from the time being extracted from literature, but an "analytical" irradiation planned within the RAPHAEL project [1], will provide data on the behaviour of fuel materials from CAPRI manufacturing facility as a function of integrated fast fluence.

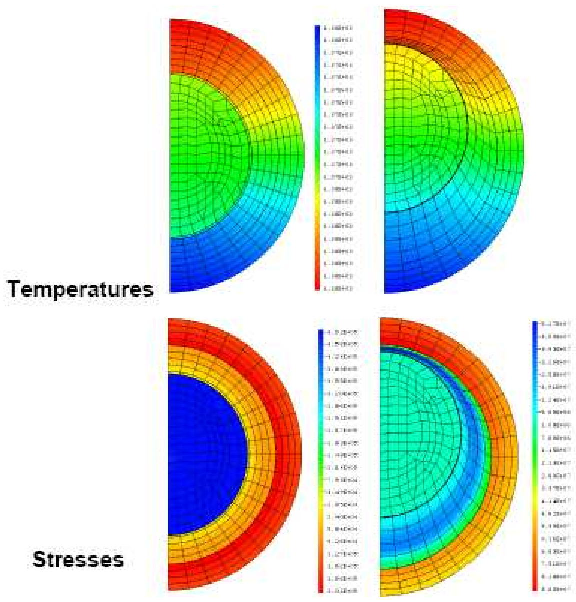


Figure 17: the modelling of the amoeba effect with the ATLAS code

For qualification of the ATLAS code, AREVA NP and CEA are participating in international code to code and code to test benchmarks in the frame of the European HTR fuel programme and of IAEA CRP6.

In particular ATLAS calculations are being compared with the results of the European irradiations and safety tests [1] and will be benchmarked with the results of the future SIROCCO irradiations.

Coupling of codes

Together with CEA, AREVA NP is involved, for the long term evolution of its design computer tools, in the development of a multi-scale, multi-physical modelling approach. In the shorter term, depending on the different time scale of the different phenomena, the code coupling needs which are the most important to implement for HTR design are those which allow iterations between reactor physics, thermo-fluid dynamics, and system analysis, with a top priority to reactor physics/CFD coupling, due to the very large temperature gradients existing in the core, which has an important feedback on the neutron cross sections. Therefore a coupling has been developed between the core design tool of NEPHTIS, CRONOS 2 and STAR-CD. Coupled core calculations show the impact of the coupling on the power and temperature distribution (figure 18).

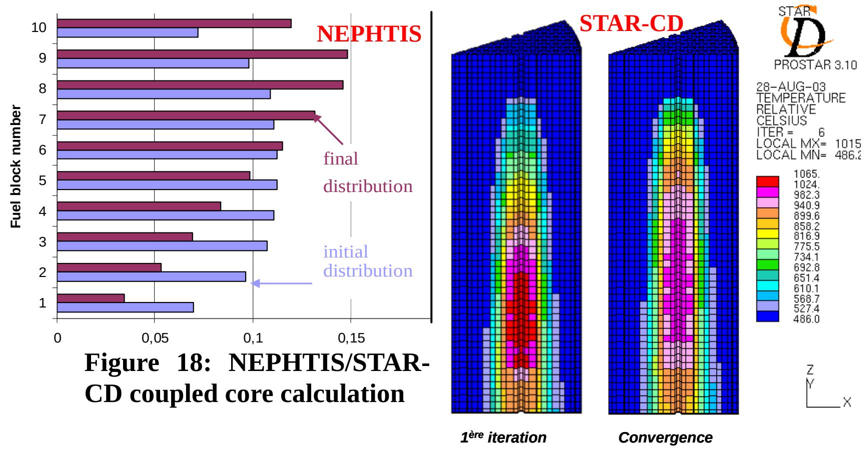


Figure 18: NEPHTIS/STAR-CD coupled core calculation

HYDROGEN PRODUCTION

The two main so-called viable hydrogen high temperature production processes that might be coupled to a HTR/VHTR are the Iodine-Sulphur thermo-chemical cycle and the High Temperature Electrolysis process. Both need to be addressed from the point of view of feasibility, optimisation, efficiency and economic evaluation for small and large scale of hydrogen production rates. Various flow sheets need to be analysed, based on different techniques for key issues. Then non nuclear material data, chemistry and thermodynamic databases need to be set up through laboratory scale tests. Performance and optimization of both processes will be assessed through integrated test loop, from laboratory scale to pilot scale before constructing a demonstration scale prototype. This will lead to the development of non nuclear component such as advanced process heat exchangers.

In parallel, other thermo-chemical cycles will be compared to the Iodine-Sulphur Thermo-chemical Cycle and High Temperature Electrolysis processes in terms of technical and economic performances in dedicated or cogeneration hydrogen production modes.

Hydrogen process coupling technology with the nuclear reactor is another key issue. The balance of plant will have to be optimised not only for mass and thermal balance, but also according to design-associated risk analysis for limiting, to the possible extent, the interfacing events between the nuclear and the non-nuclear plant, notably hydrogen explosion, tritium permeation and thermal disturbance caused by the hydrogen production system. This will require developing specific components.

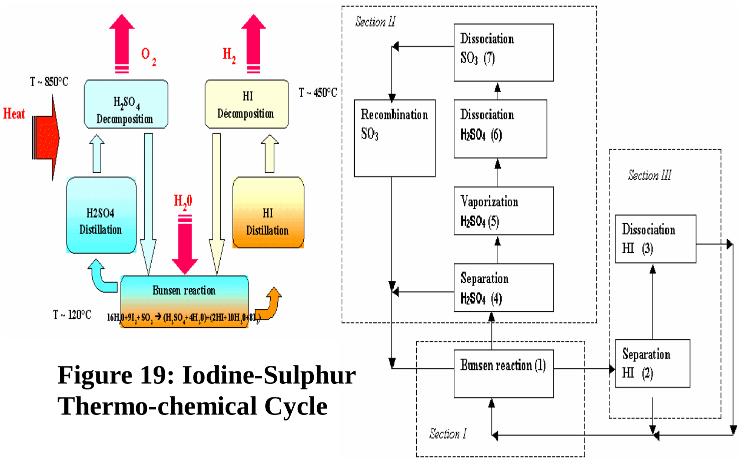


Figure 19: Iodine-Sulphur Thermo-chemical Cycle

The Iodine-Sulphur (I-S) cycle

In the assessment of the thermo-chemical I-S cycle, the overall efficiency of the process is one of the most important parameters.

Therefore, it appears essential to dedicate a strong R&D effort on the sections I and III (figure 19), to improve the models and the thermodynamic efficiency of the whole cycle [21]. Experimental data are required to optimise both HIx and Bunsen sections. The R&D programme will involve, on the one hand, design and building of appropriate reactors and, on the other hand, development of specific analytical methods to determine the composition of the liquid phases, with mainly spectroscopic techniques (UV-visible spectro-photometry, ICP-AES,…), and of the gas phases, with optical diagnostics (IR, Raman,…).

The innovative distillation-reactive concept for the $HI/I_2/H_2O$ mixture has a poor efficiency, mainly because of poor HI/H_2 conversion in vapour phase. Thus, advanced separation technologies, which include inorganic hydrogen separation membranes for efficient HI decomposition, are being studied to improve the process scheme. The mechanisms involved in the selective properties of membranes (surface reactions, pores and particles sizes), the manufacturing process of thin layers deposits shaped nano-

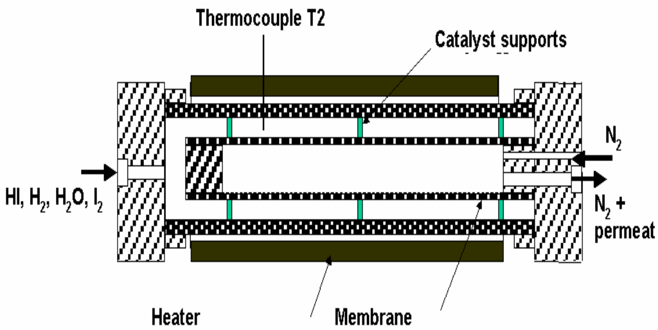


Figure 20: Membranes testing device

materials and the evaluation of the relevant methods of characterization are investigated [22]. An apparatus (figure 20) for the characterisation of the selected materials is being developed, especially to test their permeability and selectivity properties.

On the other hand, corrosion tests are necessary to assess the maximum temperature and acidity acceptable conditions, the long term behaviour and the corrosion mechanisms. Immersion tests performed in separate acids up to 140°C showed that tantalum and zirconium seem to be the most relevant metallic materials; nevertheless localised corrosion has been observed on zirconium in liquid Bunsen condition (H_2SO_4-HI-I_2 10wt%-10wt%-70wt%), (figure 21).

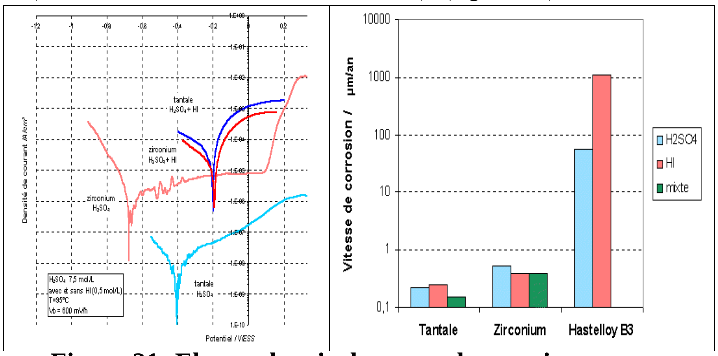


Figure 21: Electrochemical tests and corrosion rates of metallic materials

High Temperature Electrolysis (HTE)

The CEA started a programme on HTE consistent with the developments performed on the I-S cycle. It is mainly composed of developments of models for the cell (thermodynamics, kinetics, development of anode and cathode catalysts, electrode layers, micro-modelling, physico-chemical analysis) and for the system (general architecture of a stack, scale effects….) on the basis of the reversibility of a SOFC. The programme aims to the development of a HTE demonstrator, able to deliver at least 200 l H_2/h, with the Norwegian University of Science and Technology.

CONCLUSION

The R&D programme which has been defined and which is presently implemented jointly by AREVA NP, CEA and EdF is fully addressing the needs for the first phases of ANTARES design. It has already produced significant results (e.g. development or improvement of a set of computer codes which are now operational for design, with a sufficient level of validation to be confident in their results for trade studies undertaken in the present design phase for selecting the most appropriate design options, providing of key materials data, production of the first UO_2 coated particles) and it is on the verge of providing key results for the whole project, like the choice of material for the IHX. Nevertheless for later design phases and for licensing, much more results will be necessary until full qualification of the fuel, the materials, the components and the computer codes. Large test facilities, helium loops, sophisticated irradiation rigs, etc, which are presently only planned, will be necessary.

Therefore a very important R&D effort is still required for developing modern HTR/VHTR. A large part of this R&D programme is generic, common to all present projects. In order to complete it in a reasonable period of time and to make the best use of available resources, international cooperation and pooling of available research means is necessary. Such international cooperation already exists at the European level or within bilateral agreements and it has already proved to be very useful, producing relevant results. The success of the present effort undertaken by the Generation IV International Forum for widening the frame of international cooperation is a key point for the success of the whole development of HTR/VHTR.

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